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FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

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MECREDY,R.C.	Rochester Gas & Electric Corp.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 90-010-01:on 900609,reactor trip occurred from "A" steam generator caused by inadvertent closure.Caused by controller malfunction:Plant stabilize in hot shutdown & change out existing controller w/spare.W/910409 ltr.

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AREA CODE 716 546-2700

April 9, 1991

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: LER 90-010 (Revision 1), Inadvertent Closure of "A" Steam Generator Main Feedwater Regulating Valve Due to Controller Malfunction Causes a Reactor Trip On Low Steam Generator Water Level
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(iv), which requires a report of, "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System (RPS)", the attached Licensee Event Report LER 90-010 (Revision 1) is hereby submitted. This revision is necessary to revise Section III (Cause of Event) due to the root cause being identified.

Very truly yours,

Robert C Mecredy

Robert C. Mecredy

xc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna USNRC Senior Resident Inspector

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Inadvertent Closure of "A" Steam Generator Main Feedwater Regulating Valve Due to Controller Malfunction, Causes a Reactor Trip On Low Steam Generator Water Level

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 6	0 9	9 0	9 0	— 0 1 0 —	0 1	0 4	0 9	9 1		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 7

[ ] 20.402(b)	[ ] 20.405(c)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Wesley H. Backus Technical Assistant to the Operations Manager	AREA CODE 3 1 5 5 2 4 - 4 4 4 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	J B	L C	F 1 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On June 9, 1990, at 0411 EDST with the Reactor at approximately 97% full power, a reactor trip occurred from "A" Steam Generator (S/G) Low Level coincident with "A" S/G Steam Flow/Feed Flow mismatch.

The two reactor trip breakers opened as required and all shutdown and control rods inserted as designed.

The reactor trip was due to a malfunctioning "A" S/G Main Feedwater Regulating Valve Control System.

The underlying cause of the malfunctioning "A" S/G Feedwater Regulating Valve Control System was a faulty Feedwater Flow Controller. The failure of the controller was localized to a 4 Stage AC Amplifier section.

Immediate Corrective Action was to stabilize the plant in hot shutdown.

Subsequent action was to change out the existing controller with a spare.

REV1

I. **PRE-EVENT PLANT CONDITIONS**

The unit was at approximately 97% steady state full power with no major activities in progress.

II. **DESCRIPTION OF EVENT**

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o June 9, 1990, 0411 EDST: Event date and time.
- o June 9, 1990, 0411 EDST: Event discovery date and time.
- o June 9, 1990, 0411 EDST: Control Room operators verify both reactor trip breakers open and all control and shutdown rods inserted.
- o June 9, 1990, 0421 EDST: Closed both Main Steam Isolation Valves (MSIVs) to terminate plant cooldown.
- o June 9, 1990, 0431 EDST: Plant stabilized at hot shutdown.

B. EVENT:

On June 9, 1990, at 0411 EDST, with the reactor at approximately 97% full power, a reactor trip occurred. This trip was due to low level in the "A" Steam Generator (i.e. steam generator level ≤ 30%) coincident with steam flow - feed flow mismatch, (i.e. steam flow ≥ 0.8E6 lbm/hr more than feedwater flow.

The Control Room operators performed the applicable actions of Emergency Operating Procedures, E-0 (Reactor Trip or Safety Injection) and ES-0.1, (Reactor Trip Response) and stabilized the plant.

Both reactor trip breakers opened as required and all control and shutdown rods were verified inserted.

Subsequently, the Main Steam Isolation Valves (MSIVs) were closed to terminate a plant cooldown. It is believed this cooldown was partially due to cooler water being fed to the steam generators and partially due to a delay in closure of the condenser steam dump valves after the trip.

The Control Room operators notified higher supervision and the Nuclear Regulatory Commission (NRC) of the reactor protection system activation from the "A" Steam Generator (S/G) Low Level coincident with Steam Flow - Feedwater Flow (SF/FF) mismatch.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

The event was immediately apparent due to alarms and indications in the Control Room.

F. OPERATOR ACTION:

After the reactor trip, the Control Room operators performed the actions of Emergency Operating Procedures E-O, (Reactor Trip or Safety Injection) and ES-0.1, (Reactor Trip Response) and stabilized the plant. The MSIVs were closed subsequent to the trip to terminate a plant cooldown.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

G. SAFETY SYSTEM RESPONSE:

None

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip occurred due to "A" S/G Low Level $\leq$ 30%, coincident with "A" S/G SF/FF mismatch $\geq$ 0.8E6 lbm/hr.

B. INTERMEDIATE CAUSES:

The "A" S/G Low Level coincident with "A" S/G SF/FF mismatch was due to the "A" S/G Main Feedwater Regulating Valve inadvertently closing.

The "A" S/G Main Feedwater Regulating Valve inadvertently closing was due to its Foxboro Model 62H Controller (i.e. FC-466A) malfunctioning. FC-466A current output failed low. Failure low of the output current immediately causes the feedwater valve to close.

C. ROOT CAUSE:

The underlying cause of FC-466A output current failing low was the degradation of the first stage transistor in FC-466A's 4 Stage AC Amplifier. This failure was due to the change in the transistor's gain characteristics.

REV 1

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(iv), which requires reporting of "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)," in that the "A" S/G Low Level coincident with "A" S/G SF/FF mismatch reactor trip was an automatic actuation of the RPS.

of the RPS.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES: 8/31/85

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

- o The two reactor trip breakers opened as required.
- o All control and shutdown rods inserted as designed.
- o The plant was stabilized in hot shutdown.

This transient was compared to the Loss of Normal Feedwater Flow Transient described in the Ginna Updated Final Safety Analysis Report (UFSAR). None of the assumptions of the UFSAR were violated during this event.

The response of the plant to this transient is bounded by the results of the UFSAR analysis. The analysis of this transient showed that the plant responded as expected to the loss of feedwater to the "A" S/G.

During the entire event, the "A" and "B" S/Gs were always available as a heat sink due to sufficient auxiliary feedwater flow to both S/Gs and adequate steam release from both S/Gs.

Based on the above, it can be concluded that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

The Instrument and Control (I&C) Shop removed the existing "A" S/G Feedwater Controller, installed the spare feedwater controller in the "A" S/G Main Feedwater Control System, and calibrated and tested it satisfactorily.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

FCC-466A was tagged on January 25, 1991 to be discarded after the 1991 outage when the Advanced Digital Feedwater Control System is installed.

REV.1

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

The failed component was the "A" S/G Main Feedwater Regulating Valve Controller FC-466A. This controller was manufactured by Foxboro Company. The controller's model number is 62H-4E and Serial Number is 2208968.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No other documentation of similar LER events with the same root cause at Ginna Station could be identified. However, LERs 85-006, 85-019, 88-003, 88-005, and 90-007 were similar events with different root causes.

REV.1

C. SPECIAL COMMENTS:

None.